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1 January – 31 December 1980

Risø National Laboratory, DK-4000 Roskilde Denmark

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DEPARTMENT OF REACTOR TECHNOLOGY
ANNUAL PROGRESS REPORT
1 January - 31 December 1980

Abstract. The activities of the Department of Reactor Technology at Risø during 1980 are described. The work is presented in three chapters: General Information on the Department, Summary of the Department's Development during 1980, and Activities of the Department. Lists of staff, publications, computer programs, and test facilities are included.

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1. GENERAL INFORMATION ON THE DEPARTMENT OF REACTOR TECHNOLOGY

The Department of Reactor Technology consists of the following sections:

Reactor Engineering
Reactor Physics and Dynamics
Heat Transfer and Hydraulics
and the Danish Reactor No. 1

The Reactor Engineering Section has two main tasks: The first consists of following the field and collecting knowledge on the development of reactors, reactor systems, operation, economics, safety documentation, and licensing criteria. The second is the development and use of reliability methods for systems and components.

The Reactor Physics and Dynamics Section develops and utilises computer models for static and dynamic calculations on reactors, predominantly for LWRs. The models contain neutronic and thermo-hydraulic descriptions of the systems, and models are being developed for detailed core calculations as well as for overall plant descriptions.

In its theoretical work the section of Heat Transfer and Hydraulics is concerned with accident analysis, and the main task is the development of emergency core cooling models. The experimental heat transfer group is studying thermo-hydraulics, using a 0.5 MW loop. This includes stationary annular flow as well as emergency core cooling phenomena.

The Danish Reactor No. 1 is a small homogeneous water-boiler reactor used for training university students, and as a neutron source for activation analysis, predominantly neutron radiography.

The Department of Reactor Technology was formed at Risø in 1973 with the purpose of studying the subject fields described above. Other departments of Risø were dealing with metallurgy, health physics, structural mechanics, etc. The academic staff of the department working with nuclear technology numbered 32 in 1973 and increased 33% in the years up to 1977. At that time there was a diversification effort at Risø towards other types of energy research and development, and in the Department an energy systems analysis group was formed in the spring of 1977. One year later the Department began the testing station for small windmills, and later followed work on heat storage models, models, reservoir models, risk studies for oil and gas platforms, solar heating studies etc.

Today both the energy system analysis group and windmill testing station are organized at Risø outside the Department. The academic staff of the Department is 37, with 32 working predominantly on nuclear technology. The academic staff of the Department in nuclear technology represents approximately one half to two-thirds of the academic staff at Risø working on nuclear power reactor technology. A major difference from the situation at the start is that almost one-third of the staff is paid by income from contracts or are Ph.D. students.

As a consequence of the Department's development a good part of the efforts of the reactor engineering section today is directed towards non-nuclear, commercial risk studies. The reactor physicists are calculating temperature distributions and brine migration in salt depositories, and they develop models for environmental effects. The section of heat transfer and hydraulics deals with models for heat storage in aquifers and reservoirs. Solar heating using concentrating collectors is being studied both theoretically and experimentally.

The dominant effort of the Department is, however, still directed towards nuclear technology, in agreement with the objectives of Risø.

The staff of the Department is listed on page 48.

2. SUMMARY OF THE DEPARTMENT'S DEVELOPMENT DURING 1980

2.1. The Department

Whether or not nuclear power will be introduced in Denmark depends on a political decision, and before that can be taken a number of safety issues have to be evaluated by the ministries and government. One of the main issues is that of waste disposal, and here the electric utilities are reporting on the possibilities of a waste repository in rock salt in Denmark. The Department has participated in the work with calculations of actinides and brine migration. Furthermore, the Department has taken part in the work of a Danish-Swedish committee evaluating the land contamination consequences of a class 9 accident at one of the Swedish Barsebäck reactors.

Towards the end of the year the Department was being involved in work for the Ministry of Environment, which is to evaluate the safety in general of nuclear reactors. Combined with this evaluation an assessment is to be made of siting criteria for Danish sites considering large accidents and emergency planning.

The Nordic collaboration was extended during the year and through pre-projects new coordinated projects on probabilistic risk analysis and small break codes were initiated. These projects are partly financed by the Nordic Council of Ministers.

During the year the Department had a considerable amount of commercial contracts, or contracts for energy research for the government. Several of these were non-nuclear studies covering items like probabilistic risk assessment for industrial installations, and computational modelling of heat storage in aquifers.

The testing station for small windmills was transferred to another Risø department during the year.

2.2. Reactor Engineering

The work of the Reactor Engineering Section is concentrated on developing methods for assessing the reliability of systems and components in nuclear power plants. Furthermore, a core simulator is being developed.

Some years ago the work of the Section was focused strictly on nuclear power. The areas covered were of a general nature such as design, layout, and performance of modern LWR's in USA, Germany and Sweden. Also systematic studies of safety philosophy criteria and standards were performed. Due to the several postponements of the introduction of nuclear power the general scope of work was reorientated towards reliability and risk analysis. The work on nuclear power combined with methods and tools developed within the reliability fields, has proved very valuable in connection with safety and risk analysis of industrial, non-nuclear installations. Accordingly, the Section is heavily engaged in projects of this kind.

In 1979 the main projects were safety analysis of offshore oil and gas production platforms and a risk analysis of a chlorine production facility. In 1980 the transmission line in the Danish natural gas system was analysed with respect to availability of gas supply in Copenhagen, and negotiations were initiated concerning assessment of the availability of supply to the consumers. Furthermore, the Section participated in a risk analysis of a chlorine receiver station at a chemical plant. Finally, preliminary talks were started on a similar analysis of a chemical waste treatment plant; this project was, however, stopped for financial reasons.

The Section participated in an Inter-Nordic task force preparing a 4-year Nordic co-operation project on the subject "Probabilistic Risk Analysis and Licensing". The project will aim at developing tools and methods, improvement of data, and sensitivity analysis for reliability and risk analysis. Furthermore, an application of the methods developed will be tried out on an operating nuclear power plant. Finally, the requirements and limitations for the correct use of probabilistic methods in licensing will be evaluated.

Concerning the core simulator, a collaboration agreement with a Danish utility was set up. The core simulator is a system for calculating the failure probability for the individual fuel rods throughout the reactor core in light water reactors. The calculational system is set up as a modular one. The different modules are reactor physical modules and a module for calculating the failure probability for the single fuel rods.

One member of the staff was assigned to the OECD Halden experiment in Norway working on the development of methods for calculating the reliability of fuel elements. His main task has been to implement the fuel reliability code FRP on the NORD-computer and test the models used with fuel experiments. Furthermore, he is monitoring the Danish fuel elements currently being tested in the reactor.

The project on "Optimization of Reliability Techniques", which is part of a Ph.D. dissertation was completed and a final report is in preparation. A fast and efficient computer code has been designed to evaluate the probability of failure of structures using numerical integration. Furthermore, an "importance sampling" feature is being developed. This feature will be implemented in the direct Monte Carlo simulation program, MOCARE, used in the analysis of system reliability.

2.3. Reactor Physics and Dynamics

The work in static reactor physics has been concentrated on PWR-calculations. The three-dimensional, hydraulic and neutronic program for PWR cores, ANTI, has been tested further, and modifications have been made in order to make the program available for the core simulator work. The development of a new fast neutron physics calculation module based on nodal theory for ANTI has been started as part of a Ph.D. study. This module will cover both static and dynamic calculations. Another Ph.D. study concerning measurements of power distribution in reactor cores has been continued.

The ANTI program is also intended for three-dimensional dynamic calculations and has been used in connection with benchmark calculations for the Nordic Reactor Physics cooperation.

The work on plant models has been influenced by the TMI-accident. The necessity to be able to simulate extreme transients has imposed several demands on the models. A new simulation system for dynamic processes, which is used for the components in the plant models, has been made. Also a new model for a steam generator has been developed that can handle dry-out of the steam generator.

The cooperation with the Danish utilities on the SOFIE fuel management program has been continued. However, the work has been somewhat affected by the use of personal resources for qualification tests for a new computer for Risø.

In addition to the development of methods and computer programs a number of job assignments arising from outside Risø have been carried out. Work on a series of design calculations for fuel elements for the DR 3 reactor with low enrichment has been started. Furthermore, the Section has also taken part in the study of the consequences of a large hypothetical accident at the Barsebäck power plant.

The Danish utilities are evaluating a repository for highly radioactive long-lived waste in rock salt in Denmark. The Section has participated in the study with calculations of actinides in the waste, temperature distributions, and brine migration in the salt.

As a new activity, a project has been started to make a computer model, that will calculate the environmental impacts from different energy sources. The work will be carried out in cooperation with similar institutions in Finland, Norway, and Sweden.

2.4. Heat Transfer and Hydraulics

The Section of Heat Transfer and Hydraulics covers a broad spectrum of subjects in thermodynamics and hydraulics comprising both experimental and theoretical work.

In the past the Section was almost exclusively concerned with studies relevant to nuclear reactors. The last several years have been characterized by a gradual change towards the application of the accumulated experience to a number of non-nuclear subjects.

In the nuclear field the main effort has been on accident analysis. For some years the theoretical work has been performed mainly within the framework of the Nordic/American (NORHAV/LOFT) agreement. The work has been concentrated on the development of the NORCOOL-codes for the calculation of BWR emergency core cooling and the implementation and use of the US NRC code, TRAC, for PWR blowdown and emergency core-cooling calculations. In addition, some work has been done concerning "large" accidents, including core melt-down, steam explosions, etc., in an effort to keep informed about developments elsewhere.

The experimental work has included a continuation of the study of stationary annular flow in tubes and annuli, but has been concentrated mainly on emergency core cooling phenomena. The inverse annular flow regime during rewetting has been investigated in experiments both with nitrogen and water, and an experimental facility for the study of rewetting of parallel heated channels has been designed to serve as an "analog simulator" of the NORCOOL-codes.

A proposal for a study of the effect of varying pin composition on rewetting behaviour of electrically heated pins has been submitted for the "Indirect action research programme of the European Atomic Energy Community on the safety of thermal water reactors" and the project is expected to start in 1981.

An experimental determination of rock salt properties has been undertaken in connection with investigations of salt formations for the disposal of nuclear waste material.

The equipment of the temperature calibration laboratory, which was authorized in 1978, has been further improved and extended.

Solar heating for buildings has been studied theoretically and by means of small-scale experiments. Lately the work has been concerned with a preliminary study of the possibilities of utilizing concentrating solar collectors for domestic heating.

Another major effort has been concerned with the development of computer models for geothermal reservoirs and for the simulation of heat storage in aquifers. This work has been done in collaboration with the Technical University of Denmark. An extension of this work into more general reservoir modelling - including oil and gas reservoirs - has been planned for 1981.

2.5. Danish Reactor No. 1

The reactor was opened to the public during normal Risø weekend tours. This required the installation of a new automatic alarm system.

Forty nine students from technical high schools and universities were trained in a laboratory course which includes a multiplication experiment, determination of reactor constants and neutron flux distributions, and neutron radiography.

Neutron radiography using the DR 1 as source was made on several irradiated fuel pins. An improvement of radiographic images is still being attempted, and excellent radiograpns have been made using cold neutrons from the DR 3 reactor.

3. ACTIVITIES OF THE DEPARTMENT

3.1. System Reliability

A computer program package, MOCARE, was developed in the course of the latest years, for the purpose of analysing system reliability. By numerous experiences it has been proved that the application of subsystems for modelling provides an extraordinary flexibility and makes the program particularly suited to cases involving complexity in design and operation. On the other hand, since the program is based upon Monte Carlo simulation, the computational times are relatively long. A variance reduction technique will be implemented in order to circumvent this drawback, which, moreover, diminishes considerably concurrent with the development of faster computers.

A new program in the package was developed. This makes it possible to specify all input via a screen terminal, representing a very user-friendly feature of the MOCARE program package.

In some cases reliability analysis of extremely large systems by Monte Carlo simulation will be prohibitive due to the requirements for computer core and/or computer time. A new computer program was developed for analysing such systems, based upon a method developed at the Technical University of Munich. The new program selects the most important cut sets by means of Monte Carlo simulation. The calculation of the required reliability parameters for the system can then be carried out, based upon the cut sets selected, either by means of analytical methods or by means of simulation. This method however cannot solve the above-mentioned problem in cases where the cut set space cannot be adequately represented by a limited number of dominant cut sets. So far only the Monte Carlo simulation step of the method has been tested; the system under study comprised 200 components and 2×10^6 cut sets. The computation time on the Burroughs B 6700 computer was very reasonable, approximately 20 min.

3.2. Optimization of Reliability Techniques

A project, which is part of a Ph.D. dissertation, was started in 1978 with the aim of optimizing the mathematical-statistical methods used in the calculations of reliability of structures and systems.

A computer program, NUMPEP, has been developed to evaluate the probability of failure of structures based on the method of numerical integration in several variables. Some algebraic transformations are introduced to transform the original problem into an integral in several variables. The value of this integral is calculated by a numerical integration rule, based on the use of product formulas of the Gauss-Legendre or Gauss-Laguerre type.

It is possible to estimate the error introduced by the integration rule, but it requires considerable effort and the results are so conservative that they have no practical interest. In the absence of this possibility, the program is designed to calculate the integral over the entire interval and to compare this value with that obtained by dividing the integral into two equal parts, calculating these subintegrals one at a time, and finally adding them. This process can be continued on each subintegral, and it stops when the differences between two sucessive calculations is less than a specified tolerance.

This program has been compared to two others used in evaluating the probability of failure of structures. As it is shown in the table below, NUMPEP has a great advantage over ANPEP/V2 as well as the Monte Carlo simulation program PEP 706.

Several models corresponding to several types of distributions for the variables are implemented and the structure of the program is modular, so it is easy to incorporate additional models.

		Computer code		
		ANPEP/V2	PEP 706	NUMPEP
No. of	2	2.1	200.0	1.0
independent	4	12.5	400.0	1.1
variables	6	220.0	600.0	4.1
	8	2185.0	800.0	21.2

The table shows the CPU-time in seconds necessary for each code to attain a given accuracy.

Two programs, EXTEP and IMPSAMPLOT are developed, to be used in calculating system reliability. These can be used in connection with the general Monte Carlo simulation program, MOCARE.

Both programs are designed to reduce the variance of the parameters of interest in the reliability analysis, i.e. the average unavailability, the mean time to first failure, etc. Thus, variance reduction is introduced in order to reduce the CPU-time necessary to obtain a given accuracy.

EXTEP utilizes a special importance sampling technique, where the original density functions are changed in such a way that the important, but often less frequent domain of observations will be emphasized. This biasing of the simulation is corrected by weighting factors in the final result.

A certain time of observation $(0, T_{max})$ is assumed. First a failure for a component is generated between time zero and time T_{max}, say at time t_1. A repair time r_1 for this component is generated too. Now the next component has to fail before the first one is repaired. Then $t_2 < t_1+r_1$, etc. In this way, it is possible to bias the simulation so that the rare events in which we are interested, will be emphasised.

The second program, IMPSAMPLOT, is used if one has some knowledge about the behaviour of the system in advance. The technique used is again based on the importance sampling technique. In this case the program can be used for finding an importance distribution that is efficient throughout the simulation. A plotting facility is implemented to give the user an overview of the original distributions and the corresponding completed importance distributions.

These programs used in connection with the direct Monte Carlo simulation program, MOCARE, will give an efficient tool in evaluating the probability of occurrence of rare events in large complex systems

By using these techniques tests have shown that the factor of improvement over direct simulation can reach 200. The reduction is dependent on the time of observation and the size and complexity of the system.

3.3. Probabilistic Risk Analysis and Licensing Pre-Project

In preparing for a proposed Nordic co-operative project in the area of "Probabilistic Risk Analysis and Licensing" a pre-project has been carried out by an inter-Nordic task force.

The purpose of the pre-project was to establish a detailed plan for the main project and the distribution of work between the participating institutions: Studsvik Energiteknik AB, Sweden, Institutt for Energiteknikk, Norway, Statens Tekniska Forskningscentral, Finland, and Forsøgsanlæg Risø (Risø National Laboratory), Denmark. The pre-project was completed by the end of 1980.

The main project will run for four years starting in 1981 and an effort of some seven person-years per year will be put into it. The purpose of the project can be summarized in two points:

1. to provide a good measure of readiness for the performance of reliability analysis on nuclear power plant systems as far as methods, data, the understanding of uncertainties in such analyses, and value/impact analysis of alternatives is concerned, and

2. to present the conditions for and the limitations on the use of probabilistic methods in licensing.

Apart from this it is foreseen that the results of the project will be useful in designing plants and planning procedures.

3.4. Core Performance Evaluation. The Core Simulator

Operational restrictions are imposed on light water reactors in order to avoid fuel failures, or at least diminish the number of failures. As such restrictions necessarily result in reduced power production from the reactors, they are undesirable from the point of view of economics. Knowledge of the local power ramps and their consequences for the fuel is required in order to reduce the level of restrictions consistent with safety requirements.

A comprehensive system is being developed for calculating the failure probability of individual fuel rods throughout the reactor core in light water reactors. The calculational system is set up as modular. The work on the Core Simulator was begun in 1979.

One part of a Core Simulator is the calculation of the power level histories for the single fuel rods which have to be determined with an uncertainty below 10%. In order to fulfill this demand in accuracy it is necessary to consider the influence on the local power distribution caused by the boundary currents, e.g., the power distribution found in the fuel box calculation under the assumption of white boundaries cannot be used as input parameters to the fuel performance code.

A first approximation would be to use the currents found in the global 3D-nodal solution in the local flux calculation procedure.

The values of the global boundary currents depend on both the way the box average cross sections are generated and the 3D-nodal code used. At present the accuracy of the global boundary currents has been investigated and a report containing the results of this investigation is under preparation.

The above-mentioned work is part of a Ph.D. dissertation.

The other modules in the Core Simulator are the cross section module (CDB), the 3-D modules (NOTAM, ANTI) and the fuel reliability module (FRP). During the year verification of these modules has taken place, and work was started on how the coupling between modules should be achieved. The result was that a separate program should be written to facilitate the input to the single modules as well as the data transfer between modules.

A proposal for a collaborative agreement with a Danish utility has been discussed, and the next part of the work will be carried out under the terms of this agreement.

3.5 Fuel Management

SOFIE is a fuel management computer program which minimizes the fuel cycle costs for a nuclear reactor. The reactor physical treatment of the core as well as the fuel shuffling is treated by means of a 1-dimensional reactor model (concentric regions). The optimal operating strategy using this model is decided by use of linear programming with the total fuel cycle cost as the objective function.

The payment of the single components in the fuel cycle has been changed in such a way that it is possible to specify in the input to the program that the payment takes place at two time periods, earlier it was possible to specify only one point in time for the payment of the components. Thus, it is now possible, for instance, to specify that 25% of the payment of yellow cake takes place 3 years before insertion in the reactor, and that the rest (75%) takes place upon delivery of the yellow cake.

As part of the qualification tests for a new computer for Risø, SOFIE has been run on both a CDC 750 and a Borroughs 7800 computer. In order to do this a version of the program, which was written for a Borroughs computer has been converted to ANSI FORTRAN 77. As expected, results obtained on the two computers were nearly identical, however, the greater digital accuracy of the CDC computer (14 for CDC 750 compared to 11 for the B7800) proved to be beneficial for solving the linear programming problem.

3.6. Interface Methods for Solving the Neutron Diffusion Equation

It is possible to reformulate the neutron diffusion equation in several ways, so that only interface-defined quantities enter. Two such methods have been investigated earlier, one based on local Fourier-expansion in eigenfunctions of the Laplace-operator and one using integral equations connecting interface-values of flux and current. It proved difficult, with these methods, to attain the speed of programs using modern nodal techniqes. However, a number of efficient iterative methods for solution of the resulting equations were developed, in particular discretization of the equations obtained by applying inverse iteration to the original differential operator. As a possible application it is being investigated for the moment as to whether or not the running time for the MIT developed nodal program Quandry (Greenman et al., 1979) can be reduced further by means of these techniques. The equations in Quandry connect nodal average flux values and average interface currents. The flux can be easily eliminated, so that the resulting equations relate interface-defined quantities only. These equations may, in fact, be considered as being obtained by discretization of exact integral relations between point values of interface currents, so that some similarity between the numerical properties of the so-modified Quandry equations and those used in the interface method can be expected.

REFERENCE

GREENMAN, G., SMITH, K. and HENRY, A.F.(1979). "Recent advances in an analytic nodal method for static and transient reactor analysis", I-49, Proc. of the ANS topical meeting on computational methods in nuclear engineering, Williamsburg.

3.7. Three-dimensional PWR Calculations

The development of a computer model for analysing rapid transients in pressurized-water reactors was continued in 1980 and the program, ANTI, is now in the testing phase. ANTI is composed of two almost independent parts, the neutron physics part, which calculates the three-dimensional flux and power distributions, and the thermal-hydraulics part, including the model for calculating fuel rod temperatures. The power distribution from the neutron physics calculation is needed for the thermalhydraulics calculation which in turn supplies the neutronics with feedback parameters such as the fuel and moderator temperature distributions, and the moderator density distribution.

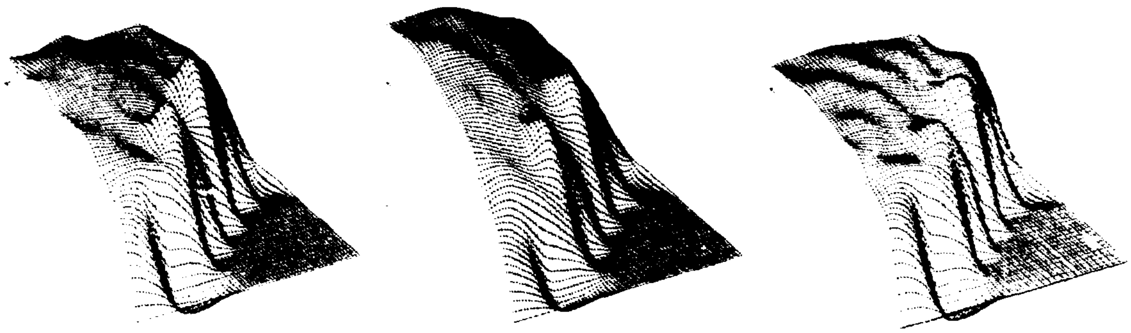
ANTI is not intended for transient calculations alone. The program is also equipped with facilities for fuel burnup calculations, and the steady-state part of the program is included in the Risø core simulator program complex, which is described elsewhere in this report. A more detailed description of the program and a user's manual is given in Nielsen and Larsen (1980).

3.7.1. PWR Static Calculations

For verifying ANTI a set of nuclear cross sections for a Westinghouse reactor similar to the Ringhals 3 design was made. For each of the different fuel assemblies the nuclear parameters were found as functions of the fuel temperature, the moderator density, and the boron concentration in the moderator.

The hydraulic part of ANTI is almost identical to the blow-down code TINA, which has been subject to extensive tests previously. Therefore, most of the effort in the verification of ANTI was given to the flux and power calculation methods. The neutron physics part of ANTI is based on the TRILUX nodal model and the probabilities for neutron passage across the internodal boundaries are given as functions of the nodal dimensions, the local cross sections, and four empirical parameters g_i, $i = 1.4$. The flux and power distributions are strongly dependent on the choice of g-parameters. Compared with Westinghouse calculations and calculations with the two-dimensional finite difference program TWODIM at Risø, it was concluded that different power conditions can be modelled rather well with the same g-set. However, if the nodal configuration is changed then g_4 should be adjusted. These analyses are fully described in Thorlaksen (1980) and demonstrated in Fig. 1, where the horizontal power distribution is shown for one quarter of a PWR core.

From the investigations above it was concluded that the flux solution method in ANTI can be used. However, the need for a better method is still felt, and a Ph.D. study has been initiated in this area.



Case 1	Case 2	Case 3
One nodal channel per fuel assembly	Four nodal channels per fuel assembly	Four nodal channels per fuel assembly
$g_4 = 0.65$	$g_4 = 0.65$	$g_4 = 3.25$

Fig. 1. Horizontal power distributions for different nodal configurations and coupling parameters. The two nodal configuration cases 1 and 3 represent best fits compared with TWODIM. Note that in case 2, in which g_4 has not been fitted to the specific node size, the power is more peaked in the central region than in the two other cases.

3.7.2. PWR Dynamics Calculations

As an example of the three-dimensional dynamics calculation for a PWR, Fig. 2 shows the calculated development of the horizontal power distribution during the initial part of a control rod ejection transient. The example is a three-dimensional version of a one-dimensional benchmark calculation proposed by Finland for the Nordic Reactor Physics meeting in November 1980. The one-dimensional ANTI calculation is in reasonable agreement with the Finnish solution by the TRAWA program; however the one-dimensional representation of the reactor core contains information only about the axial distribution of the horizontally averaged power, and provides no clue to the changes in power shape illustrated by the figures.

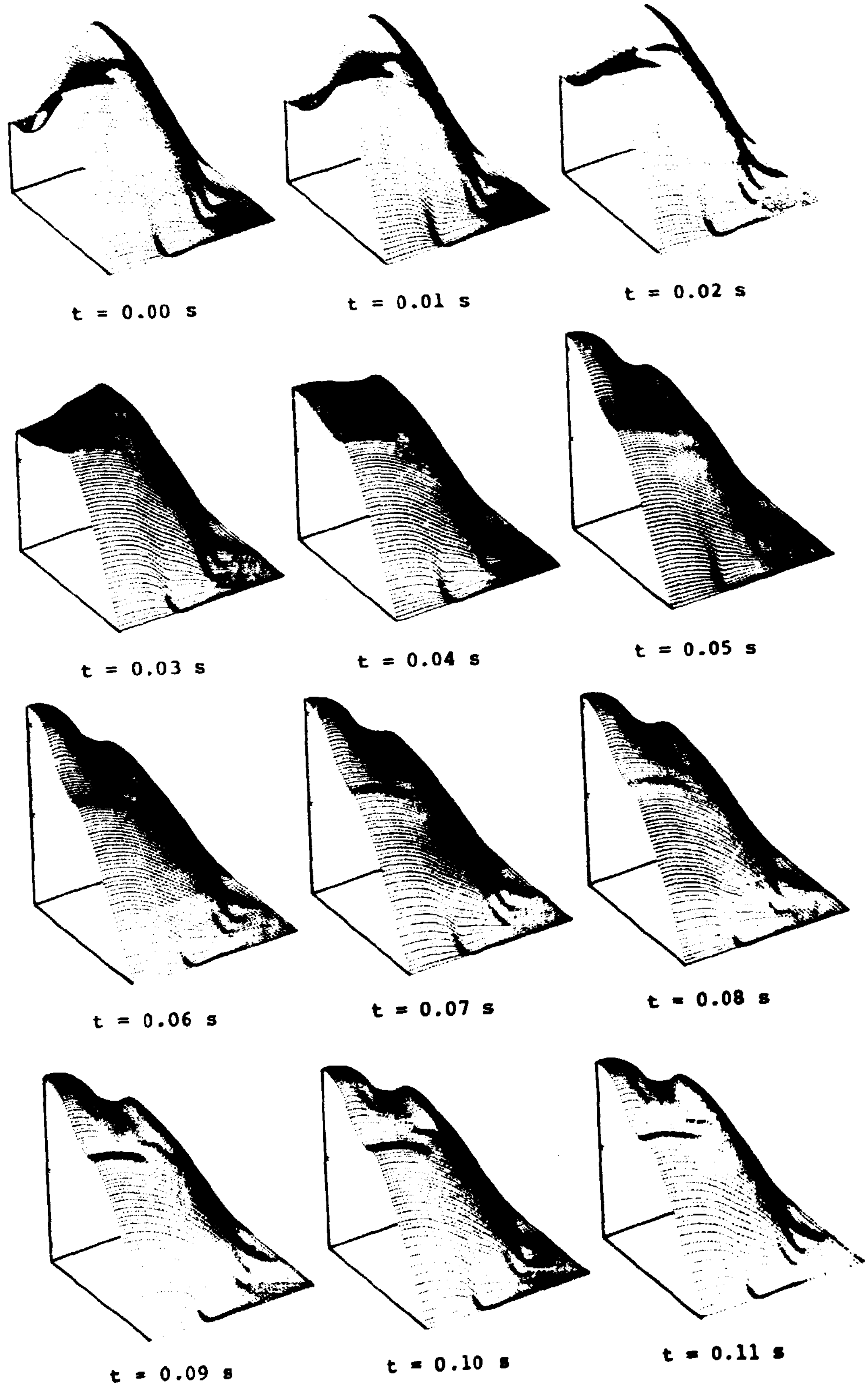


Fig. 2. Development of the normalized horizontal power distribution for a quarter of the core during the first 0.11 second of a contro' rod ejection calculation. The control rod initially inserted in the center of the core is fully withdrawn at the time 0.05 second.

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NIELSEN, E.F. and LARSEN, A.M.H. (1980). Input Description for the Three-Dimensional PWR Transient Code ANTI. Risø-M-2256.

THORLAKSEN, B., (1980). Construction of PWR Nuclear Cross Sections for Transient Calculations. Test of the ANTI Program against TWODIM. Risø-M-2264.

3.8. Development of a Simulation System DYSIM for Continuous Dynamic Processes

In 1975 a program DYSYS for simulating dynamic processes was bought from the Kernforschungszentrum Karlsruhe (Schlechtendahl, 1970). In the past five years it has been used extensively for simulating nuclear reactors and plant components. Several minor modifications have been made to adapt the program to our computer and our special demands. Major modifications could not be made due to the structure of the program, which is very complicated and difficult to work out. Therefore, it was finally decided to write a completely new program based upon the experience gained in the work with DYSYS.

For the many models which have been developed for DYSYS there has been a major demand that the old models described by FORTRAN subroutine and input data files should be usable with only minor modifications. So the new program, called DYSIM, has been developed along the same lines as DYSYS. But some features are not used in DYSIM, and new ones have been introduced. The accuracy control and steady-state search have been completely changed and the routines most essential for the computing time have been programmed carefully in order to save time at the expense of memory space. The time performance is improved by a factor of 1.4-1.5 for typical examples, and the program is cut down from about 1900 FORTRAN lines to about 1000.

The main features are:

- The program is written in FORTRAN and includes the main program and a number of subroutines. It takes care of data input with test and documentation, integration of system equations, and organisation of outputs. Several administration procedures are included.

- The problem equations must be formulated in FORTRAN subroutine DERIVA, which gives the derivatives for the first-order differential equations.

- The structure of the model may be altered so differential equations are changed to algebraic ones and back again at any time during the calculations.

- The integration routine is a fourth-order Runge-Kutta.

- Steady-state calculations can be performed and the result stored in a disk file as initial values for the next run.

- During the transient calculation, accuracy control with step-size adaption is carried out in a simple and fast manner.

- Output tabulation is specified in the input file and the selected values are stored intermediately in a disk file, which may be used to make plots later on.

- Pure time delays can be simulated by means of a separate function, DELAY, which must be included in the user program when it is needed.

- A dump and restart facility provides the possibility of dumping the state of the system in a disk file and in a later run restart the calculation at that state. This is useful for test purposes when an error occurs long into a transient.

DYSIM has been tested by simulating typical power plant transients and is now operational. It will also be used for simulating BWR and PWR påower plants including one-dimensional reactor models and models for pressurizers, steam generators, steam turbines, and feedwater heaters.

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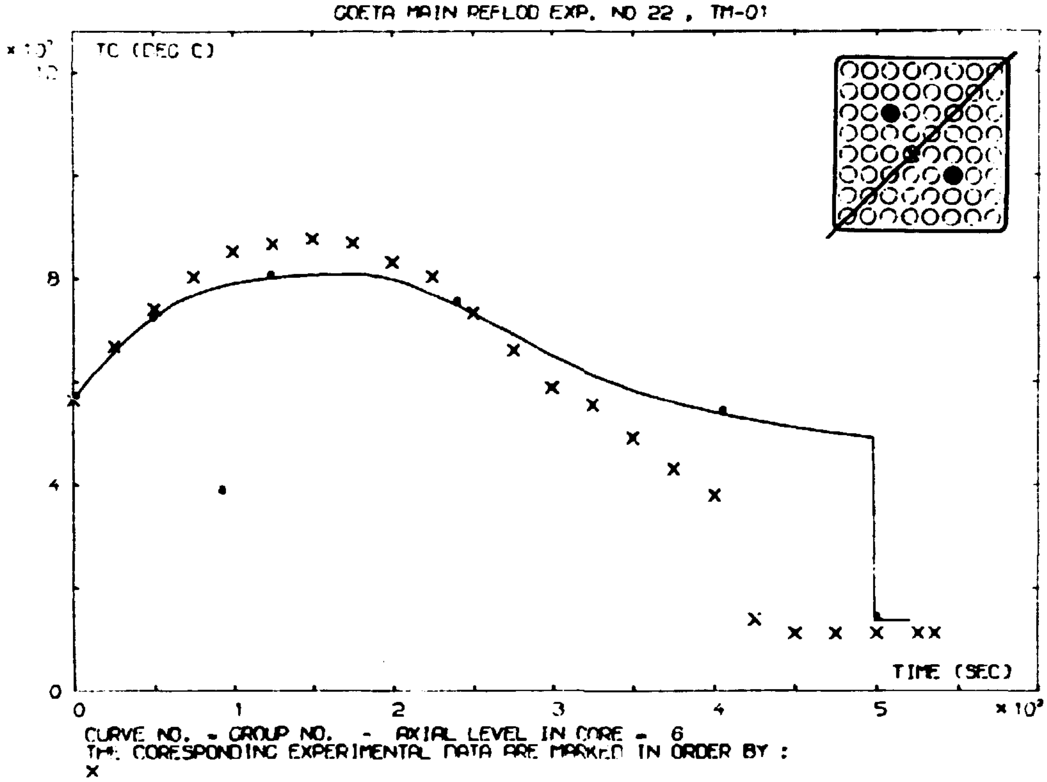
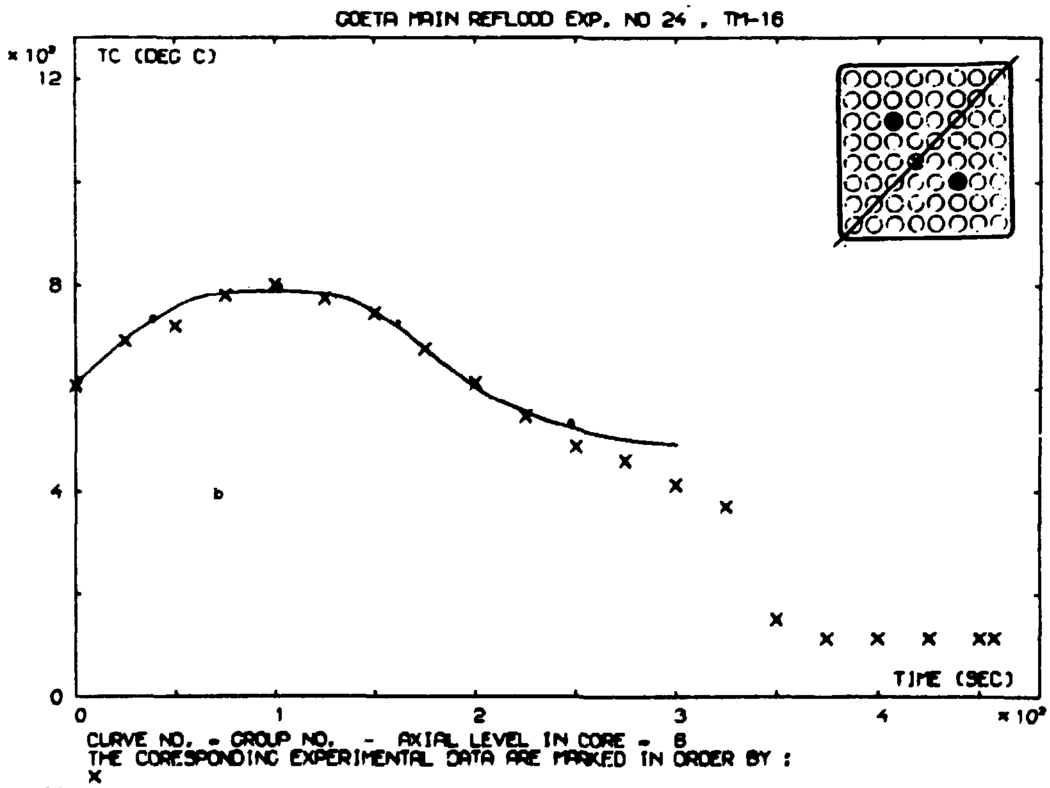
SCHLECHTENDAHL, E.G. (July 1970). DYSYS - A Dynamic System Simulator for Continuous and Discrete Changes of State, Institut für Reaktorentwiklung, Karlsruhe, KFK 1209.

3.9. The BWR Emergency Core Cooling Program NORCOOL-I

NORCOOL-I is a BWR emergency core cooling program able to model both the spray and reflood cooling following a loss of coolant accident (LOCA). The geometry representation is fixed to a scaled BWR vessel geometry, consisting of one fuel element and accordingly scaled volumes for lower plenum, by-pass, upper plenum, steam dome, and downcomer. The break can be placed anywhere in the downcomer volume. In the two-phase flow calculation the three water fields: falling film, droplets, and continuous water, and the two gas fields: continuous gas and bubbles, are all treated separately, thus allowing unequal temperatures and velocities for the different fields. In USA the program is incorporated in a program package called WRAP-EM, meaning "Water Reactor Analysis Program Evaluation Model", in which input to NORCOOL-I is generated from the output of a RELAP4 calculation of the blowdown phase of the LOCA. In April 1980 an updated version of NORCOOL-I was sent to USA to replace the current version in WRAP.

Few changes have been made to the physics and numerics of the program during 1980. A few stability problems were solved at the beginning of the year and some inconsistent treatments of the fuel element box when compared with the fuel rod treatment were corrected in the late autumn.





24-14

Fig. 3. NORCOOL-I calculations compared with experimental results.

Rather many corrections have been made to the input and especially to the output parts of the code to reduce the possibility of input errors and to give a more consistent output.

Parallel to this work a new user's manual, i.e. input/output description, has been written, and the work on a new model description has been started. The program has been tested against ten of the Swedish Göta-main-experiments, a series of spray and reflood cooling tests of a BWR fuel element. Nine of the ten runs showed almost the same discrepancies between the calculated and the measured cladding temperatures, (see Fig. 3[a]), while a single run gave much better agreement (see Fig. 3[b]). So far we have not been able to explain the difference in behaviour in this test when compared with the other nine.

3.10. The Advanced BWR Emergency Core Cooling Program NORCOOL-II

NORCOOL-II uses non-equilibrium three-field hydraulics (steam, water film, and water droplets) and it relies basically on a one-dimensional geometric representation of the various parts of the network. For the core a number of parallel fuel channels are used as illustrated in Fig. 4.

In 1980 the work proceeded on three fronts: 1) testing and running-in of the network mechanism, 2) final programming and test of a heat component module, and 3) reporting.

The working of the hydraulics in the network geometry was tested using as test cases a) water level oscillations in a U-tube and b) water injection in a steam-filled 3-channel reactor-like system. The occurrence in the calculations of unphysical pressure jumps, when calculational cell boundaries were passed by a water level, made it necessary to introduce a new bubble friction model and to implement a more efficient interpolation of the steam-water interface friction between the bubble and the annular flow regimes.

The above tests were performed with suppressed occurrence of water droplets. However, after adequate models for steam-droplet interface friction and heat transfer, and for droplet entrainment (i.e. tearing out of droplets from a water film by the steam flow) was implemented, allowance was made for droplets. A repeated run of the 3-channel reactor-like test case showed that the hydraulics for steam, water film, and water droplets worked satisfactorily.

During the testing of the hydraulics as described above the wall temperature was calculated by a preliminary, rough wall heat conduction model.

However, in parallel with this testing the programming of the so-called heat component module was finished. This module performs the detailed calculation of the heat conduction in the fuel rods and other components bounding the flow channels including the calculation of the movement of quench fronts. The heat component module was tested separately and was found to work satisfactorily. In a couple of idealized test cases with radial and axial conduction, respectively, for which an analytical solution exists, the agreement with the analytical solution was perfect.

After the heat component module had been implemented in the main program a single test run of the 3-channel test case showed that the heat component module and the hydraulic part of NORCOOL-II functioned well together.

During the last part of 1980 the testing and development work with NORCOOL-II have been resting, and the effort has been concentrated on reporting NORCOOL-II at its present stage. During this reporting a number of ideas and plans for the future development of the code has been listed. Among these ideas is an improved model for the so-called added-mass force on a bubble for situations when the volume fraction of the bubbles is comparable with that of the water.

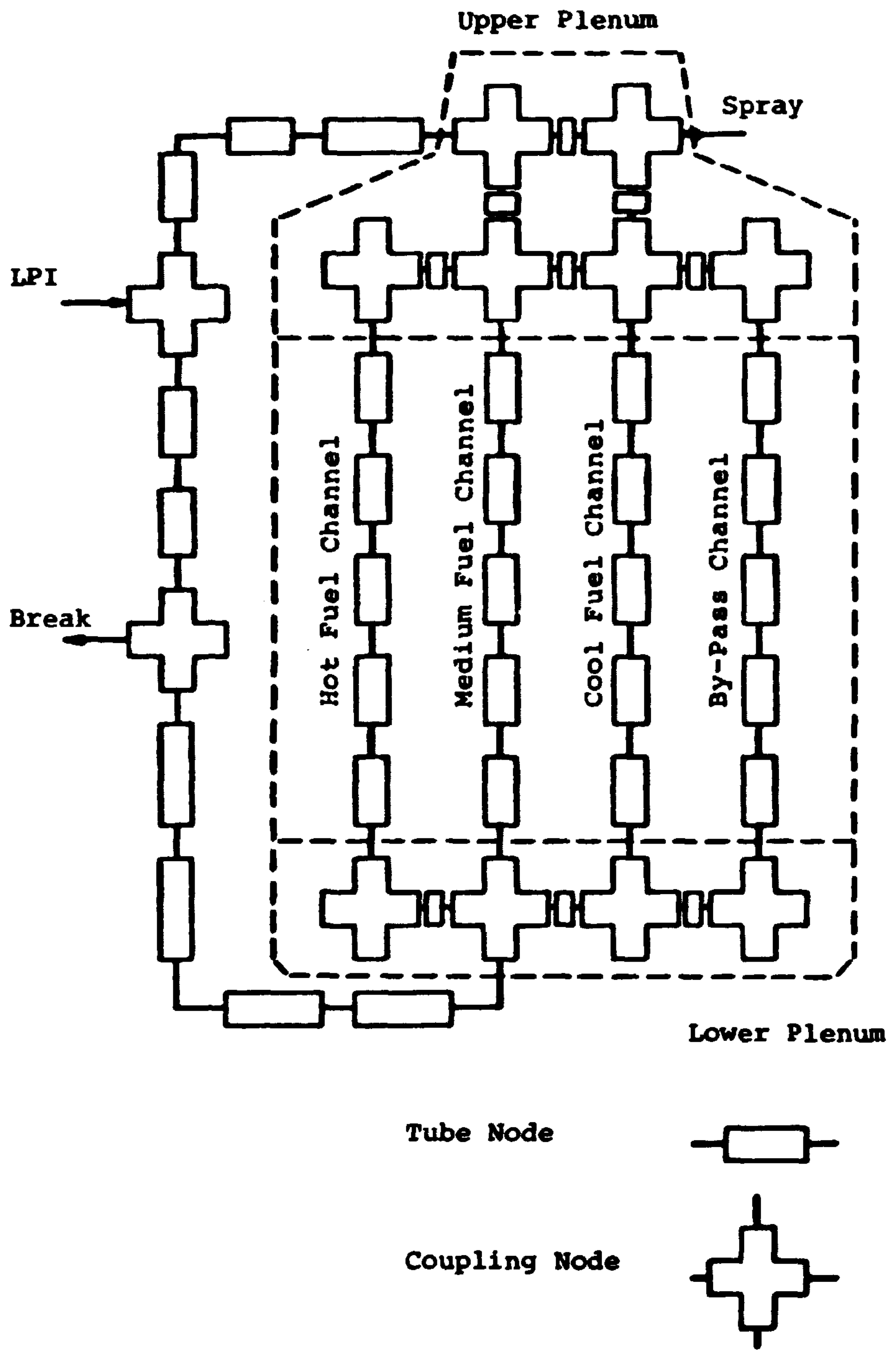


Fig. 4. Example of NORCOOL-II Network Representation of BWR Core Geometry.

3.11. The PWR LOCA Analysis Code TRAC

TRAC-PD2, the version which replaces P1A, has been received from *Los Alamos Scientific Laboratory* (LASL). The major improvement are in the areas of reflood and heat transfer models, interfacial relations, and numerical methods. TRAC-PD2 will be implemented either on the Cyber 175 at CDC Stockholm or on a Cyper 176 at CDC in Brussels. Implementation of the new TRAC version at Risø's computer is pending the decision concerning the acquisition of a new computer.

3.12. Participation in the LOBI Project Prediction Exercise

The first test of the LOBI experimental program A1 has been used for a blind prediction exercise (PREX). Sixteen different organizations from member countries of the European Communities, and from the USA participated in the exercise, in which about forty different quantities had to be predicted.

In the PREX test a double-ended cold leg rupture of 2 x 100% break size was simulated. The core power stayed constant for about 1.8 s at 100% and was then dropped to zero. The accumulator with the emergency core cooling water was connected to the cold leg and the pressurizer to the hot leg of the intact loop. The actual initial and boundary conditions of the experiment were communicated to all participants.

Most of the participants used the RELAP4 code. In two cases, however, the more advanced TRAC-P1A code was applied (LASL and Risø). The TRAC predictions are characterized by a depressurization that was too fast caused by a critical mass flow rate that was too large during the saturated period of the blowdown.

The following conclusions can be drawn from the comparison between calculations and measurements:

1) The break-flow calculation is crucial.
2) User experience is crucial, especially in the case of a code such as RELAP4 with numerous options and dials.
3) The primitive best estimate code RELAP4 gives more improved results than TRAC-P1A. The main reasons are:
   - The experiment is relatively simple and its description does not require sophisticated mathematical models.
   - Considerable experience has been accumulated concerning the application of RELAP4 to similar problems (e.g. Semiscale)
4) The best RELAP4 predictions are good.

3.13. Inverse Annular Film Boiling Experiments

Inverse annular film boiling may occur during emergency core cooling of a nuclear reactor. The present experiments are part of a Ph.D. study, which aimed at a better understanding of this boiling regime. The Ph.D. study was finished at the end of June.

The main part of the Ph.D. work consisted in performing steady-state experiments in a glass test section, where the inverse annular regime could be kept stationary with nitrogen as a model fluid. For different flow rates and nitrogen subcoolings the heat transfer and the axial void fraction profile were measured. The final analysis of the experimental data, performed with the thermohydraulic two-fluid code RISQUE (equipped with nitrogen data), showed excellent agreement between data and RISQUE predictions when relevant heat transfer correlations were used. This result was presented at the European Two-Phase Group Meeting, June 3-6 in Glasgow.

The introductory part of similar experiments with water in a stainless steel test section was also performed before the end of the Ph.D. work. However, it proved very difficult, if not impossible, with the present design of the test section to obtain a stationary inverse annular regime. This observation raised the question as to whether or not the inverse annular regime occurs at all during emergency core cooling except for the very first part of it. However, a poor design of the test section is just as likely to be the reason as any other for the missing observation of a stationary inverse annular regime.

The entire Ph.D. work has been reported, and the work will be presented at a public lecture in January 1981.

3.14. Experimental Simulation of NORCOOL-I and II

An experimental facility is being designed with the purpose of simulating the geometry for the loss of coolant analysis, which is done by the NORCOOL I and II codes. The experiment has four parallel channels, two simulating the fuel, one the downcomer, and one the by-pass.

3.15. Experimental Study of Rewetting and Quench Phenomena

A proposal has been made for participating in the "Indirect action research programme of the European Atomic Energy Agency Community on the safety of thermal water reactors".

The aim of the project is to study the effect of varying pin compositions on the rewetting and quench behaviour of the pins under a series of identical boundary conditions.

For this purpose it is proposed to make experiments using electrically heated pin simulators ranging from directly heated pins via conventional indirectly heated pins to advanced indirectly heated pins containing ring pellets of uranium dioxide. These last-mentioned ones simulate nuclear-heated pins more closely.

An analysis of experiments of this type may help make more reliable predictions of rewet and quench behaviour in nuclear reactors under accident conditions.

The proposal has been negotiated with the European Community and is unanimously supported by the experts subgroup and the

Advisory Committee for the Management of the Reactor Safety Programme to obtain EC contribution.

The project is expected to start in spring 1981 with a planned duration of 3 years.

3.16. Temperature Calibration Laboratory

The Temperature Calibration Laboratory was authorized in 1978 by the Danish National Testing Board to carry out certified calibration of temperature sensors in the temperature range from -150°C to 1100°C. The standard thermometers in the laboratory are traceable to National Physical Laboratory, England.

In 1980 the Laboratory has performed 44 jobs for external customers and 8 more for other departments of Risø. These used almost 200 thermometers of different kinds ranging from conventional liquid-in-glass instruments to advanced electronic digital thermometers. The calibrations have been made in the temperature range from -90°C to 1100°C.

The equipment in the Laboratory has been improved, with a new cryostat and a new electrical furnace of the Laboratory's own design and construction. The measurement equipment has been extended with a photocell galvanometer amplifier and a spare standard 25Ω resistance thermometer.

3.17. Thermal Properties of Rock Salt

In connection with investigations of disposal of radioactive fuel waste in salt formations, measurements of thermal properties have been carried out such as thermal conductivity, specific heat, and emission factor by samples of rock salt from a Danish saltdome.

The measured values of thermal conductivity and specific heat show good agreement with values from the literature. On the other hand it has not been possible to find values of the emission factor of rock salt from the literature. The emission factor was measured by means of an infrared thermometer which measures the radiation intensity in the wavelength range of λ = 8-14 µm. The emission factor was found to be very close to that of a black body.

3.18. Brine Migration

As part of a feasibility study of storing highly radioactive waste in Danish salt deposits carried out by Danish power utilities, Risø has investigated problems concerning brine migration. Small inclusions of brine (droplets of water) will usually be found in rock salt. These have a tendency to travel along temperature gradients, thus causing of brine water to be transported towards the heat-producing waste. As the water may corrode waste canisters, models have been made to calculate the amount of water which can be transported to the canisters.

The project is now terminated and calculational models developed in the preceding year have been tested against experiments carried out under the American project Salt Vault.

In order to estimate the influence of the special geometry characteristic of this particular experiment a method of evaluating the total brine inflow to a given heat source has been developed, which is independent of the exact geometrical shape of the source. The method has been derived from general considerations of the coupling between the temperature field and the field of brine density.

The agreement between experiment and calculation is fair considering a number of uncertainties regarding the parameters of the experiment. Also a disagreement between calculations and reality of a factor of 1.5, which is found in this study, would probably be acceptable for design calculations.

3.19. Solar Heating of Buildings

Solar heating systems for buildings are studied theoretically and supplemented by small-scale experiments.

The possibility of using large concentrating collectors for solar heating systems under Danish climatic conditions is being investigated in a pre-project supported by The Ministry of Energy. This work is divided into the following three parts:

1. A preliminary design of a prototype solar collector is in progress. The collector is of the dish type with a diameter of the order of 5 m. The focusing accuracy is not required to be as high as for high-temperature collectors. Furthermore, the collector is supposed to be protected against wind and precipitation by being placed under a transparent dome or roof. Thus a light low-cost construction is made possible.

2. Measurement of direct and global solar radiation. The data are obtained as ten-minute average values stored on magnetic tape.

3. The good thermal performance of a focusing collector makes it particularly suitable for applications with seasonal storage. Both individual and collective systems are investigated. Based on the Danish standard reference year the collector output is estimated to be about 700 kWh/m^2yr. A more accurate evaluation including the system heat losses will be performed.

In order to obtain some operational experience a test collector with a 1.5 m-diameter reflector has been built. This collector is expected to be used for testing various heat transfer techniques.

3.20. Seasonal Heat Storage in Aquifers

Since 1978 Risø has participated in a joint project on seasonal heat storage in aquifers. In collaboration with the Laboratory of Energetics at the Technical University of Denmark the development of numerical models to simulate the mass and energy transport in porous media is being carried out. During 1980 the developed, two-dimensional, finite element model D2AQ has been applied to the design of the 50,000-m^3 pilot plant at Hørsholm.

One of the important results achieved during this investigation is that it seems possible to control the flow in the storage by using a combination of fully and partially penetrating wells, and hence to some extent counteract the unfavourable warm and cold water mixing caused by buoyancy effects.

The calculations have been carried out in the vertical plane bounded by the center well, one of the four boundary wells, and the upper and lower impermeable layers. Figure 5 gives examples of calculated temperature distributions and flow fields during the storage and production periods.

During the year a development of the model has been carried out, which also enables it to calculate the pressure and temperature field in the horizontal plane.

On the basis of these two two-dimensional versions the development of a fully three-dimensional simulation model is now inititiated.

3.21. Geothermal Energy

During 1980 the Department participated in two projects concerning geothermal energy. Both were performed under contracts with Dansk Olie og Naturgas A/S.

The first project was an economic evaluation of possible sitings of geothermal district heating plants with respect to geological as well as market factors. In this connection a computer code has been developed, which made it possible to simulate geothermal energy sources where the heat flux between the geothermal water and the water used for district heating partly passes heat pumps. The heat pump technique seems to be a necessary contribution to the utilization of low temperature geothermal resources.

The second project had the purpose of estimating the possible geothermal resources in this country. This report was made in collaboration with the Danish Geological Survey and the University of Aarhus.

In the report the country is subdivided into four areas, where the reservoir characteristics within each area are maintained constant. In all four areas the calculations have been made for two permeabilities, an estimated maximum and an estimated minimum permeability.

The coefficient of performance, i.e. the heat that can be delivered for district heating versus the amount of invested primary energy, has been calculated for the eight different cases. It was necessary to include some economic consideration, and in figure 6 a typical connection is shown between the coefficient of performance and the internal interest.

3.22. Environmental Effects of Energy Production

Every kind af energy production will have an impact on the environment. Some are rather obvious, such as smoke from chimneys, ash, slag etc. However, in order to get a background for comparing the impacts from different energy sources or to assess the combined effects of several energy sources in an energy system a systematic effort has to be made.

Work has been started on a computer model which should be able to represent an energy system and to calculate the environmental effects of the system. The model should consider all parts of the system, that is:

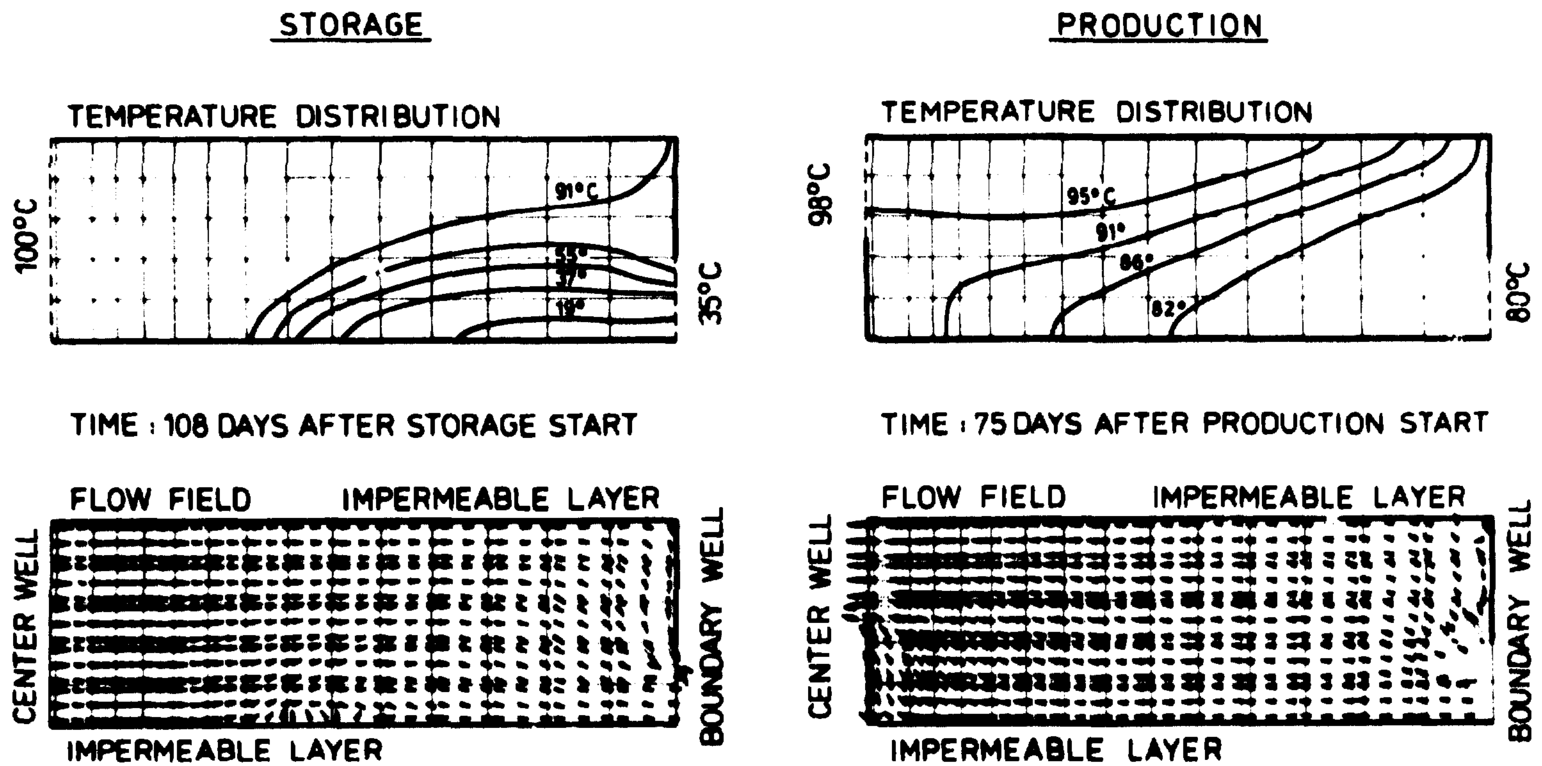


Fig.5. Calculated temperature distributions and flow fields in an aquifer storage. During the storage period the center well fully penetrats the aquifer, whereas only the upper half is open during the production period. The boundary wells are open only at the lowest third of the aquifer. The figure indicates that this well configuration - in comparison with a design with fully penetrating wells - delays the breakthrough times of the injected water, and hence increases storage efficiency.

The calculations have been carried out with the following data: Volume flow rate: 25 m^3/h. Initial temperature storage period: 10°C. Aquifer thickness: 15 m. Distance between center well and boundary well: 40 m. Permeability: 10 Darcy.

- mining of fuel,
- transport of fuel,
- construction and operation of power plants,
- removal of waste, and
- distribution of the produced energy.

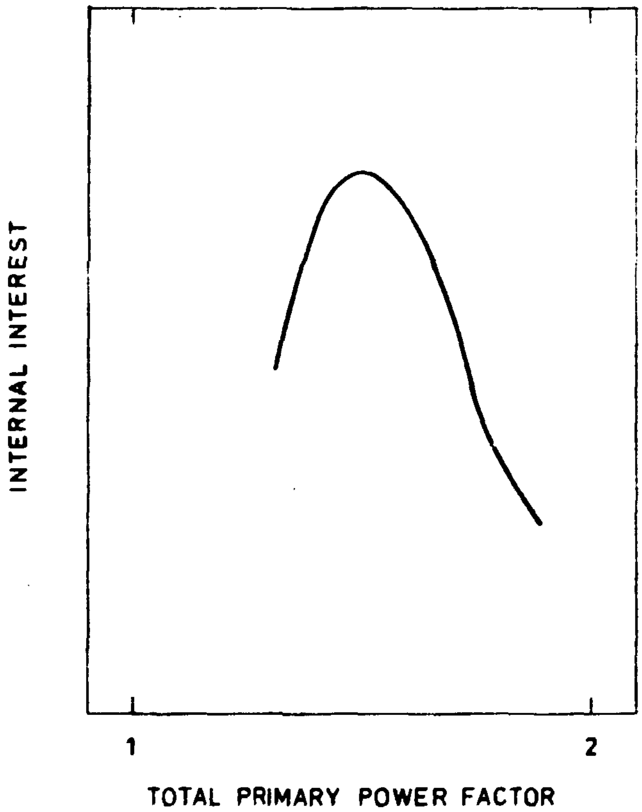


Fig. 6 shows a typical calculation of the internal interest of an investment in a geothermal power plant versus the total primary power factor of the plant. The plant is expected to work for a period of 20 years. Both the internal interest and the power factor are dependent of the flow of geothermal water, and it is therefore possible to decide on a size of plant that gives the maximum internal interest. The power factor of the plant can be determined from the figure. In this case it is about 1.5.

Ideally the final goal of the model would be to calculate such effects as:

- occupational hazards,
- impacts on ecological systems,

- health effects,
- material effects (e.g. corrosion), and
- changes in climate.

The emphasis will not centre on creating sophisticated mathematical models, but rather on building a model that can handle available data. In many areas such data do not exist and will hardly come into existence within a reasonable time, as for instance the influence of any energy production on the climate The model should be able to assess the impact on the environment as far as existing knowledge makes it possible and may help to identify, in a systematic way, areas in which further data is needed.

As coal is a widely used energy source it was chosen to provide an energy system model. As a first step a model containing submodels for

- transport of fuel,
- power plants,
- atmospheric dispersion, and
- terrestic ecology

will be made.

These submodels interact through a main program in which a description can be given of the geographical region which is to be studied for environmental effects. The submodels for transport and power plants will essentially be simple input-output models. The dispersion model will be based on a data file calculated by existing atmospheric dispersion models and the ecology model will be based on simple transfer factors. In later stages of the project it is the intention to follow the pollutants further through the different biological pathways. In this connection, models for the aquatic ecology have to be added.

To demonstrate the use of the model a scenario has been defined. The Danish island of Zealand on which Copenhagen is situated, and which has five coal-fired power plants sites, is chosen as the region to be considered.

In the last part of the project, data for consequences, e.g. health effects are to be introduced in order to estimate the impact of pollution. It is the purpose of the work to extend the model and data to several energy systems and to compare various energy alternatives.

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STAFF OF THE DEPARTMENT OF REACTOR TECHNOLOGY, 1. March 1981

Head: B. Micheelsen

Office staff:

N. Christensen
I. Kristiansen
G. Egelund
D. Solomon
I. Strandvad

Reactor Engineering

Head: P.E. Becher

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K. Alstrup

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C.F. Højerup
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Programmers:

E.F. Nielsen
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* On leave at Halden Reactor Project, Halden, Norway

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Heat Transfer and Hydraulics

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DR 1 Reactor

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Scientific staff:
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Technical staff:
B.K. Hansen
K.C. Nyvang Nielsen

APPENDIX A

COMPUTER PROGRAMS

Description	Code name
Reliability Analysis	
System Reliability Monte Carlo simulation with or without variance reduction. Very flexible modelling based upon "subsystems". User-friendly interactive input code is available, and a graphical display of component- and system performance can be obtained.	MOCARE
Component Reliability Calculates the probability of failure of a structure for models used in probabilistic fracture mechanics. The calculations are based upon numerical integration in one or several dimensions.	ANPEP
Reactor Physics. Data Processing System	
Group Cross Section Generation Generates 76-group master tape from UKNPL.	SIGMA
Resonance Data Generates self shielded few group cross sections in the resonance region using collision probability calculations in several thousand groups.	RESAB
Scattering Data Generates multi-group scattering cross section based on the NELKIN model.	NELKINSCM

Reactor Physics. Fuel Pin and Element Calculation

Pin and Cluster Cell Calculation 76-group, collision probability, pin cell calculation. Collision probability cluster cell calculation and burn up calculation in arbitrary number of groups.	CCC
LWR Fuel Element Calculation Multi-group, collision probability pin cell calculation with burn up. XY-diffusion theory calculation for element.	CDB

Steady State Reactor Physics, Diffusion Theory, Overall Reactivity, Flux and Power Calculation

Two-dimensional Difference Equation	TWODIM TVEDIM
Two-dimensional Finite Element	FEM
Three-dimensional Difference Equation	DC4
Three-dimensional Finite Element	FEM
Three-dimensional Flux Synthesis	SYNTRON
BWR Three-dimensional Nodal Theory Multi-Channel one-dimensional flow hydraulics	NOTAM
PWR Three-dimensional Nodal Theory Subchannel hydraulics	ANTI
Dynamics General program system for simulating dynamic processes.	DYSYS

Three-dimensional Dynamics

BWR Dynamic Model — DANAID

Nodal theory neutronics. Multichannel, one-dimensional flow hydraulics.

PWR Dynamic Model — ANTI

Nodal theory neutronics. Subchannel model hydraulics.

Plant Dynamics

BWR Model — BWR-PLASIM

One-dimensional reactor model coupled to one-dimensional steam line model, turbine model and feed water system, including control options.

PWR Model — PWR-PLASIM

One-dimensional reactor model coupled to model for primary cooling system with pressurizers and one-dimensional steam generator model. Secondary side with one-dimensional steam line model, turbine model and feed water system, including control options.

BWR Model with Detailed Turbine Model — BWR PLANT

One-dimensional finite element neutronics, one-dimensional hydraulics for the reactor, coupled to detailed turbine model.

Shielding — SHIELDING

Attenuation of γ radiation from a cylindrical volume- or surface-source. Cylindrical or slab shields in multiple layers.

Fuel Management — SOFIE

The program minimizes the fuel cycle costs for a nuclear reactor. The reactor physics as well as the fuel shuffling is treated by a one-dimensional model.

Reactor Steady State Heat Transfer and Hydraulics — SDS

Subchannel core heat transfer and hydraulics. Best-estimate, steady state, flow and enthalpy, BWR and PWR, non-equilibrium, drift-flux, boundary value solution technique.

Reactor Accident Analysis

PWR Blowdown — TINA

Core, best-estimate, subchannel approach, non-equilibrium, drift-flux.

BWR Top Spray Emergency Core Cooling — CORECOOL

Best-estimate, spray cooling, one-dimensional, non-equilibrium two-fluid plus falling films, detailed radiation heat transfer.

BWR Emergency Core Cooling — NORCOOL-I

Best-estimate, spray cooling and reflooding, one-dimensional, saturated steam drift-flux in continuous water regions, non-equilibrium two-fluid plus falling film in continuous steam regions, explicit two-phase level and quench front tracking, detailed radiation heat transfer.

PWR Blowdown and Emergency Core Cooling — TRAC

Best-estimate, LOCA system analysis, three-dimensional and one-dimensional, non-equilibrium, two-fluid and drift-flux, bottom flooding and falling film.
(Program from Los Alamos, USA)

PS-Containment LOCA Response — CONTAC-III

Two-room plus vessel, quasi-stationary, thermal equilibrium.

Containment Response to Core Meltdown — MARCH

Vessel melt-through, interaction of core debris with water and concrete and hydrogen burning. (Program from Battelle Columbus, USA).

Calculation of Fission Product Transport and Deposition — CORRAL

Containment systems of water-cooled reactors. The containment thermal hydraulics required for this program is provided by MARCH.
(Program from Battelle Columbus, USA)

Geothermal Energy and Heat Storage in Aquifers

Two-dimensional, Linear Finite Elements — PORFLOW

Simplified temperature field represented by a hot and a cold zone, x-y and r-z geometry.

Two-dimensional, Quadratic Isoparametric Finite Elements — D2AQ

Detailed temperature field, x-y and r-z geometry.
(Program from Laboratory for Energetics. Technical University of Denmark).

Heat Pumps and Geothermal Heat — GEOPOWER

Two-dimensional, analytical reservoir model, detailed desctiption of heat pump circuits including coolant parameters.

Ground Water Pollution Dispersion — SWIP

Three-dimensional, finite differences, pressure, energy- and mass-transfer, well-bore calculation, x-y-z and r-z geometry.
(Program from U.S. Geological Survey).

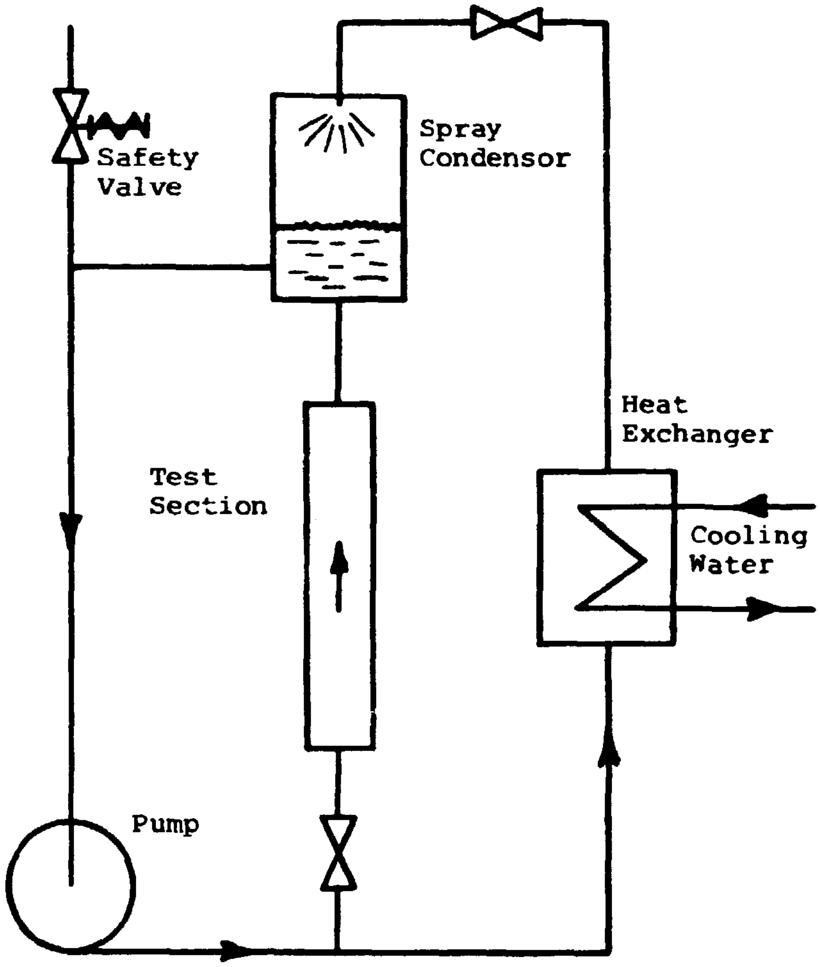
APPENDIX B

TEST FACILITIES

Danish Reactor No. 1

Type	Aqueous-homogeneous
Thermal output	2 kW
Temperature	25°C
Fuel	Uranyl sulphate
Enrichment	20%
Reflector	Graphite
Max. neutron flux	
thermal	$6 \cdot 10^{10}$ n/cm^2 s
fast	$12 \cdot 10^{10}$ "
Experimental positions	1 thermal column
	1 through tube
	8 positions in reflector

High Pressure Water Loop



Pressure	221 bar
Temperature	375°C
Flow	10 l/s
Test section	
Length	9 m
El-power	660 kW DC

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